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Point Beach Nuclear Plant
Operated by Nuclear Management Company, LLC

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Resolution of Safety-Related Questions Regarding
Unit 2 Reactor Vessel Head Lift

In accordance with discussions held between representatives of the Nuclear Regulatory Commission (NRC) and Nuclear Management Company, LLC (NMC) on May 6, 2005, Enclosure 1 to this letter provides information in response to specific questions regarding the reactor vessel head (RVH) lift of Unit 2. Specific areas discussed include mitigating strategies and dose consequences.

Enclosure 2 provides a probabilistic risk assessment that demonstrates the upper bounding scenario for core damage probability is less than 1E-6 occurrences per lift.

Summary of Commitments

There are no new commitments or revisions to existing commitments in this letter.

NMC remains confident that the mitigating strategies and analyses described within the Enclosures, confirms our ability to safely replace the Unit 2 reactor vessel head. NMC's defense in depth approach demonstrates that NMC will continue to provide a reasonable assurance of protecting the health and safety of the public during RVH lift activities.

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Enclosures (2)

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ENCLOSURE 1

RESOLUTION OF SAFETY-RELATED QUESTIONS REGARDING UNIT 2 REACTOR VESSEL HEAD LIFT

Information contained in this enclosure responds to questions raised during discussions between representatives of NRC and NMC on May 6, 2005.

Mitigating Strategies for Maintaining a Coolable Core Geometry

The upper bounding scenario for dropping the RVH onto the reactor vessel is that severe damage will occur to piping attached to the reactor coolant system (RCS) and the ability to remove decay heat by normal means will be lost.

At the time of the planned lift, more than 38 days will have elapsed since reactor shutdown, and one-third of the core has been replaced with unirradiated fuel. As a result, the total decay heat load will be approximately 5.5E+6 BTU/hr, and the requirements for makeup due to decay heat boil-off will be less than 12 gpm.

With this heatup rate and assuming an initial temperature of 100°F, there will be at least 15 hours before the volume of water remaining in the reactor vessel heats to the point of boiling. This heatup rate provides sufficient time to implement the mitigating strategies before the onset of boiling.

Prior to suspending the Unit 2 RVH over the vessel, NMC will install a temporary modification that will provide two redundant, temporary connections to the RVH. These connections will be made to the reactor vessel level indication system penetration and to the reactor head vent penetration, providing a means to inject borated water from the refueling water storage tank (RWST) onto the vessel upper internals. Each of the connections through the head are ¾" Schedule 160 pipe (nominal inner diameter [ID] of 0.612"). This small bore piping is short and constitutes the main flow restriction in the lines.

The lines will be supplied from each of the two redundant containment spray headers to the RVH, which operate at a minimum of 40 psig above containment design pressure (60 psig). With a 100 psig supply pressure, each of these lines would deliver an adequate flow to accommodate core boil-off. Prior to lifting the RVH, it will be verified by calculation that either line is capable of delivering water flow in excess of the core boil-off requirements.

The temporary lines connections will be made up and verified aligned to the containment spray headers prior to suspending the RVH over the reactor vessel. In the event of a RVH drop event that causes severe damage to the RCS, manual initiation of either train of containment spray will ensure adequate core cooling is maintained to remove decay heat and keep the core covered. By installing these temporary lines, the core damage probability is reduced from 5.6E-5 to 4.7E-7 per lift. Enclosure 2 provides an assessment of core damage probability.

Upon exhaustion of the RWST inventory, the residual heat removal pumps may be realigned to take suction from the containment sump with the containment spray pump(s) drawing from the residual heat removal pump discharges. This provides assurance that the containment and core spray can be maintained for a prolonged period.

The containment spray system is borated (drawing suction from the RWST), ensuring that dilution does not reduce the shutdown margin. Long-term concerns with boron concentration and precipitation can be managed by switching to demineralized (DI) water later if warranted. Note that with the calculated boil-off rates, such a condition would not be expected for many days post-event.

Steam from boil-off will be vented via a combination of the three open core exit thermocouple RVH penetrations and the RVH/vessel flange gap[1].

In addition, prior to positioning the RVH over the vessel, a substantial steel block will be placed on the vessel mating surface. In the event of an uncontrolled RVH drop, this block will prevent the RVH from sitting squarely on the vessel, and provide a gap through which temporary cooling water may be supplied directly into the vessel. The block designated for this purpose is an approximate 8" diameter, 6.5" tall cylinder of 4140 steel. The block will be removed after the head height above the flange has been reduced to 24" or less, and prior to final RVH set.

As an upper bounding scenario, all six upper vessel penetrations are considered completely lost such that no water can be injected via normal paths, and the reactor vessel drops to the containment base mat at Elevation –1'. Several (or all) BMI penetrations are faulted, potentially to the point of severance.

In this scenario, the top of active fuel will then be at approximately Elevation 18.5' and the core mid-plane would rest at approximately plant Elevation 12.5'. Injection of the entire contents of the RWST into containment (through a combination of containment spray, residual heat removal, and/or safety injection) will flood the containment to approximately Elevation 13.7', a little more than one foot (1') above the core mid-plane.

[1] Due to the combination of both the upper internals circumferential spring and the fuel assembly hold-down springs, the weight of the RVH assembly at PBNP is insufficient to seat the RVH on the vessel flange unless the closure studs are tensioned. Impact of the RVH on the vessel flange will momentarily compress these springs to their normal installed configuration. The impact is not expected to cause damage to the springs or internal components. Once the impact energy has been absorbed, the springs will rebound and restore the RVH/vessel flange gap.

Until containment flooding increases water level to the top of the active core, the previously described temporary RVH injection system would provide cooling flow necessary to prevent core damage.

Discharging against ambient pressure, the two trains of available residual heat removal, safety injection and containment spray pumps can transfer the entire RWST inventory into the containment in approximately 40 minutes.

Since some leakage may exist beneath the reactor vessel, additional water will need to be injected into containment to raise the level to the top of the core and ensure long-term submergence of the entire core. To raise the containment level an additional five feet (5') to reach the top of active fuel will require approximately 228,000 gallons of additional water. This volume is achievable using existing plant procedures and inventories (275,000 gallons of borated water available in the opposite unit's RWST, boric acid storage tanks, waste holdup tanks, etc.).

Dose Assessment

The assessment of dose consequence was performed assuming an instantaneous gap release into containment equivalent to two fuel assemblies based on information contained in Westinghouse Nuclear Safety Advisory Letter (NSAL) 04-6. This letter states that Westinghouse has reevaluated the rod cluster control assembly (RCCA) drive rod buckling loads for a spectrum of original RVH assembly weights, replacement RVHs, and RVH assembly upgrade packages. For Westinghouse-supplied fuel with Westinghouse-supplied drive rods, Westinghouse determined that based upon new calculations of drive rod buckling loads, the fuel assembly structure may sustain damage, but fuel rod cladding integrity would be maintained. Therefore, no fuel rod cladding damage is expected due to the concentric drop of the reactor vessel head onto the vessel. Since operation is permitted with 1% fuel defects, which is the equivalent of 1.2 assemblies, it is conservatively assumed that due to impact of the reactor vessel head on the vessel, an instantaneous release equivalent to the gap activity contained in two assemblies is released.

Airflow measurements were conducted on May 7, 2005, that physically validated containment air outflow previously been based on theoretical flows due to natural ventilation.

No other activity is expected to be released since the core will be maintained cooled and there is no additional damage postulated.

The dose assessment uses the following assumptions:

- Equivalent of two (2) fuel assemblies gap activity is released;
- Instantaneous release rate from fuel to containment atmosphere;
- Purge isolates in 15 seconds to account for valve closure;
- No radial peaking factor applied because no actual fuel clad damage postulated to occur.

The estimated offsite doses due to an instantaneous release of the gap inventory of two fuel assemblies are as follows:

	Exclusion Area Boundary	Low Population Zone
30 minutes, without purge release	0.18 rem TEDE	0.01 rem TEDE
30 minutes plus filtered Purge release	0.2 rem TEDE	0.011 rem TEDE
60 minutes, without purge release	0.37 rem TEDE	0.02 rem TEDE
60 minutes plus filtered purge release	0.57 rem TEDE	0.031 rem TEDE
CLB FHA (NRC SER dated April 2, 2004)	1.6 rem TEDE	0.1 rem TEDE

The current licensing basis fuel handling accident (FHA) is a two-hour duration release from containment based upon an instantaneous release from the fuel. Therefore, at one-hour following the licensing basis fuel assembly accident, the dose is 0.8 Rem TEDE that bounds the postulated consequences of the heavy load drop. The estimated release at one-hour post-heavy load drop is well-within the 10 CFR 100 limits. The control room dose would also be bound by the current licensing basis FHA. No reliance on the ingestion of potassium iodide (KI) is assumed.

Summary

Although containment is expected to be isolated within 30 minutes after a RVH drop event, this assessment provides reasonable assurance that if additional time is needed to obtain containment closure, protection of the health and safety of the public would continue to be assured.

ENCLOSURE 2

PROBABILISTIC RISK ASSESSMENT TO ESTIMATE CORE DAMAGE PROBABILITY

A probabilistic risk assessment was performed to estimate the core damage probability associated with the lift of the new RVH over the reactor vessel. The estimate considered the probability of dropping the RV head along with the Conditional Core Damage Probability if the head were to drop. For this estimate it was assumed that the only core injection method available would be via hose connections from the containment spray system to the RVH as described in Enclosure 1. This is considered the limiting case because all others have multiple and diverse flow paths and equipment available.

Initiating Event

The initiating event in this assessment is the drop of the RVH while it is suspended over the reactor vessel. The RVH is assumed to fall onto the reactor vessel flange, resulting in damage to the attached piping such that normal injection methods (safety injection, residual heat removal and charging) are not available.

NUREG-1774, "A Survey of Crane Operating Experience at U.S. Nuclear Power Plants from 1968 through 2002," was written to address NRC Candidate Generic Issue 186, "Potential Risk and Consequences of Heavy Load Drops in Nuclear Power Plants." Crane operating history from 1968 through 2002 was reviewed as part of this report to provide a risk assessment associated with lifts of Very Heavy Loads (VHL). The risk analysis included in NUREG-1774 considers VHL lifts for any crane at any operating nuclear station. The analysis considers a postulated drop of load at any point during the movement of a load from the initial lift until set-down. In addition, the risk assessment included in NUREG-1774 was set-up to determine the probability of a number of different end states (consequences).

The probabilistic analysis contained within NUREG-1774 is primarily concerned with the probability of a VHL drop at an operating commercial nuclear power plant. A VHL is defined as any load over 30 tons. The generic probability for any VHL drop is given as 5.6E-5 per lift. This value is based upon three (3) drops in 54,000 VHL lifts.

A plant-specific review has been performed to demonstrate that operational characteristics with respect to crane failures due to mechanical failures or human performance are not significantly different than the average of plants considered within NUREG-1774. Three (3) areas were reviewed and compared to the generic data included within NUREG-1774.

- VHL Drop Probability – PBNP data review indicates that approximately 429 VHL lifts were performed using the turbine building, primary auxiliary building or containment cranes between the period of January 1, 1995, and April 14, 2005 (average of approximately 20 per reactor per year). There were no drops identified during this time. NUREG-1774 provides a probabilistic value of 5.6E-5 per lift. Statistically, given the small number of VHL lifts performed at PBNP, it is not expected that a drop event would have occurred. This review suggests that PBNP does not deviate from the values contained within NUREG-1774.

- Load Slip Probability – Plant review indicates 429 VHL lifts. During this timeframe, PBNP experienced zero (0) VHL load slips. NUREG-1774 states that there were six (6) load slips in the 54,000 VHL lifts considered. This review suggests that PBNP does not deviate from the values contained within NUREG-1774.

- Human Error Probability (probability of human error per lift) – A review of plant data shows there were 29 human error and procedural events out of 50 lift-related events in more than 14,000 lifts of any size that took place between January 1, 1995, and April 14, 2005. The majority of the events are human error and procedure related. This is similar to the observation noted in NUREG-1774, demonstrating that PBNP is not an outlier compared to the data contained in this assessment.

All three drops referenced within the NUREG-1774 involve a failure of rigging and all involved a human error associated with the rigging. It is considered that this value is conservative and bounding for an RVH drop for the following reasons:

1. If a load were to drop as a result of a rigging problem, there is a likelihood that the load drop will happen at the beginning of the lift because of the lift rig failing when it is first put under stress. For this RVH lift, there is some likelihood that the load drop may occur when the RVH is not suspended over the reactor vessel, or that it occurs from a low height. During the RVH set, only the end of the lift takes place over the reactor vessel. If rigging failure occurs during the RVH installation, it is much more likely to occur at the beginning of the lift when the RVH is not above the reactor vessel. Because these split fractions are not known with great certainty, it is assumed for this assessment that any drop that occurs takes place over the reactor vessel.

2. The three VHL drops cited in NUREG-1774 were all failures of nylon or Kevlar sling-type riggings being used on cranes not located in containment. These rigging failures were, at least in part, attributed to human error resulting in the slings being overstressed or unprotected from damage during the lift. The rigging used for the RVH lift is constructed of steel, is specifically designed for this lift and is used exclusively for this lift. The RVH rigging and crane is inspected prior to the lift. The RVH lift is rigorously controlled by procedure, and key personnel involved are experienced with this particular lift.

Considering the factors discussed above, it can be stated with a high degree of confidence that the PBNP plant-specific probability of a RV head drop is less than the upper bound estimate of 5.6E-5 per lift provided in NUREG-1774. The three VHL drops that have occurred in the industry were attributed to a failure mode that cannot occur for a RVH lift because a single purpose, steel lifting rig is used rather than a general use, nylon or Kevlar sling. Even though these three failures can be eliminated because the specific failure mode does not apply to a RVH lift, the entire population of 54,000 successful VHL lifts can be used because all of the remaining possible failure modes are still applicable to all of these VHL lifts.

With these three VHL drops eliminated, the correct number of failures for the numerator is now some value between 0 and 1. It is a common PRA practice in the situation where no failures have occurred to use an estimated value of 0.5 in the numerator. Assuming 0.5 drops in a sample size of 54,000 VHL lifts results in a more appropriate VHL drop probability of 9.3E-06 per lift.

For this assessment, a bounding drop probability of 5.6E-5 is assumed, but based upon the above discussion, it is believed to be conservative by a factor of six (6).

Conditional Core Damage Probability

The estimated Conditional Core Damage Probability is based upon the plant PRA model for the failure probability of both trains of containment spray. The model was adjusted to account for potential human errors that may occur due to the specific initiating event being postulated. The Conditional Core Damage Probability may consist of any of three failures: (1) Failure to initiate containment spray; (2) Failure to establish sump recirculation after draining the refueling water storage tank (RWST); (3) Equipment failure associated with the containment spray system, residual heat removal system and all support systems. The human error associated with containment sump recirculation was assumed to be bounded by the evaluation for a large loss-of-coolant accident (LOCA), which requires recirculation early in the event, and assumes a high stress level. Equipment failures were evaluated by solving the current plant PRA model. A Human Error Probability (HEP) to account for failure to initiate containment spray when necessary was estimated based upon specific procedures and training provided for this event.

The HEP estimate for the initiation of containment spray is based upon the manual action to start at least one train of containment spray prior to core damage. A combination of EPRI Cause-Based Decision Tree Method (CBDTM) and Technique for Human Error Rate Prediction (THERP) methods was used. The scenario evaluated starts with control room notification from the field of a dropped RVH followed by entry into the Abnormal Operating Procedure (AOP) for this event. Credit is taken for the initiation of containment spray upon verification of the event and the inability to manage core inventory through the appropriate shutdown LOCA procedures. The actions necessary to start containment spray are simple with all controls available from the control room. The analysis estimates a failure probability of 1.4E-3.

Considering the three basic failure modes discussed above, the overall Conditional Core Damage Probability was determined to be approximately 8.4E-3.

Core Damage Probability

The probability of a RVH drop times the Conditional Core Damage Probability provides the Core Damage Probability per RVH lift. Using the bounding RVH drop probability of 5.6E-5 and a Conditional Core Damage Probability of 8.4E-3, it is estimated that the Core Damage Probability is 4.7E-7 per lift. Results demonstrate that the upper boundary scenario for core damage probability is less than 1E-6 and the dose consequences are well within allowable limits.